

ACCESSION #: 9910200198
NON-PUBLIC?: N

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Browns Ferry Nuclear Plant Unit 2 PAGE: 1 OF 5

DOCKET NUMBER: 05000260

TITLE: Manual Reactor Scram due to an EHC leak

EVENT DATE: 09/15/99 LER #: 1999-009-000 REPORT DATE: 10/14/99

OTHER FACILITIES INVOLVED: NA DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

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COMPONENT FAILURE DESCRIPTION:

CAUSE: E SYSTEM: TG COMPONENT: TBG MANUFACTURER: G080
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On September 15, 1999, at 1825 CDT, Unit 2 operators manually scrambled the reactor from 54 percent power due to an Electro-Hydraulic Control (EHC) [TG] leak that could not be isolated. The reactor had been at 100 percent power prior to the leak. As expected, the reactor scram caused reactor water level to go below the low level setpoint (level 3) which generated a redundant scram signal and initiated Primary Containment Isolation, Standby Gas Treatment, and Control Room Emergency Ventilation Systems. All systems responded as expected and all control rods fully inserted.

The cause of the leak was failure of a stainless steel tubing connection that was installed for the power uprate modification package to measure pressure perturbations in the EHC system. The damaged tubing was removed and the connection plugged.

TVA is reporting this event in accordance with 10 CFR 50.73 (a)(2)(iv) as an event that resulted in a manual actuation of an engineered safety feature, including the reactor protection system.

END OF ABSTRACT

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I. PLANT CONDITIONS

Prior to the initiation of the event, Unit 2 and Unit 3 were at 100 percent power. Unit 1 was shutdown and defueled.

II. DESCRIPTION OF EVENT

A. Event:

On September 15, 1999, Unit 2 developed an Electro-Hydraulic Control (EHC) [TG] leak that could not be isolated, upon recognition of the problem, the operators took appropriate action by reducing reactor power and manually scrambling the reactor prior to an automatic scram from a turbine trip. The reactor was initially at 100 percent power prior to the leak and was scrambled from 54 percent power after initial operator action. As expected, the reactor scram caused reactor water level to go below the low level setpoint (level 3) which generated a redundant scram signal and initiated a Primary Containment Isolation, Standby Gas Treatment, and Control Room Emergency Ventilation Systems. All systems responded as expected and all control rods fully inserted.

The cause of the leak was failure of a stainless steel tubing connection that was installed for the power uprate modification package to measure pressure perturbations in the EHC system.

The scram resulted in the expected automatic actuation or isolation of the following PCIS [JE] systems and components:

- o PCIS group 2, Shutdown cooling mode of Residual Heat Removal (RHR) [BO] system; drywell floor drain isolation valves; drywell equipment drain isolation valves [WP].
- o PCIS group 3, Reactor Water Cleanup (RWCU) system [CE].
- o PCIS group 6, primary containment purge and ventilation [JM], Unit 2 reactor zone ventilation [VB]; refuel zone ventilation [VA]; Standby Gas Treatment system [131-1]; Control Room Emergency Ventilation system [VI].
- o PCIS group 8, Traversing Incore Probe (TIP) [IG].

This event is reportable in accordance with 10 CFR 50.73 (a)(2)(iv), as an event that resulted in a manual actuation of an engineered safety feature, including the reactor protection system.

B. Inoperable Structures, Components, or Systems that Contributed to the Event:

None.

C. Dates and Approximate Times of Major Occurrences:

September 15, 1999, at 1758 hours CDT	An EHC Reservoir Level Low alarm was received in the control room and personnel were dispatched to investigate.
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C. Dates and Approximate Times of Major Occurrences (continued):

September 15, 1999, at 1804 hours CDT	Reactor power was lowered to 50-60% core flow upon receipt of report that the EHC reservoir level was reported low.
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September 15, 1999, at 1825 hours CDT Reactor manually scrammed.
Expected PCIS signals and
actuators occurred when reactor
water reached level 3 following
the scram.

September 15, 1999, at 1844 hours CDT A four-hour non-emergency report
is made to the NRC pursuant to 10
CFR 50.72 (b) (2) (ii).

D. Other Systems or Secondary Functions Affected:

None.

E. Method of Discovery:

Operators received alarms indicating an EHC leak had occurred.

F. Operator Actions:

Operations personnel responded to the event in accordance with applicable
plant procedures.

G. Safety System Response:

All required safety systems operated as designed.

III. CAUSE OF THE EVENT

A. Immediate Cause:

The immediate cause of this event was failure of a stainless steel tubing
connection in the heat affected zone of the weld.

B. Root Cause:

The root cause of the failure was poor fabrication and work practices used
to install the stainless steel tubing.

C. Contributing Factors:

None.

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IV. ANALYSIS OF THE EVENT

As part of the installation for five percent power uprate during the Browns
Ferry Unit 2 Cycle 10 refueling outage, a design change installed four EHC
accumulator packages on the main turbine control valves to dampen EHC
pressure perturbations. Part of this installation package included a 3/8
inch nominal outer diameter (0.035 inch nominal wall thickness) tubing
connection which consisted of socket weld glands and standard nuts to
connect the accumulator to a pressure transmitter on the number four main
turbine control valve. This stainless steel tubing connection completely
fractured at the toe of the weld and resulted in the necessity to initiate
a manual scram.

The subject tubing failure was evaluated in order to determine the failure
mechanism and root cause for the failure. Plant personnel that initially

discovered the failed EHC tubing failure indicated that the broken segments of the tubing were off-set at least 2-3 inches. This amount of offset would result in excessive cold springing for tubing of this diameter and length. A visual examination performed by Site Engineering on the inside of the tubing at the failure location showed evidence of weld melt-through at the root of the joint for almost the entire circumference of the tubing. The weld melt-through is the result of excessive heat input from welding during fabrication of the failed socket weld joint. Examination of the tubing fracture surfaces using a stereo microscope revealed a relatively flat fracture surface at the toe of the weld, ratchet marks and the absence of gross deformation. Scanning electron microscopy, which was performed at TVA's Central Laboratories, on the tubing side of the fracture surface revealed striations.

The features (i.e., relatively flat fracture surface, ratchet marks, striations and the absence of gross deformation) revealed by stereo and scanning electron microscopy on the fracture surface of the stainless steel tubing failure are indicative of a high cycle fatigue failure. This connection was exposed to constant vibration during plant operation. The excessive cold springing and weld melt-through resulted in additional residual stresses which attributed to this failure. Therefore, the root cause of this tubing failure is poor fabrication and installation practices.

No other similar installations were identified on Unit 2 or Unit 3.

V. ASSESSMENT OF SAFETY CONSEQUENCES

The evaluation of plant system and component responses to the event concluded that responses were as designed and within the time-frames expected. The normal heat removal path was not lost during this event since the condenser was used for decay heat removal and no main steam relief valves opened. Personnel performance was also evaluated and found to be timely, appropriate, and met expectations for performance during an event of this type.

There were no equipment failures during or following the scram that complicated recovery. In addition, there were no radioactive material released and no actual or potential safety consequences as a result of this event. Therefore, this event did not adversely affect the safety of plant personnel or the public.

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VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions:

The Operations crew stabilized the reactor following the scram using the appropriate operating instructions.

The failed tubing was removed and the connections were plugged.

An inspection of the area affected by the EHC fluid was performed and cleanup activities were completed prior to restart.

B. Corrective Action to Prevent Recurrence:

General Electric will evaluate the cause of failure and provide recommendations to TVA to prevent recurrence.1_/

TVA will evaluate the design of the tubing and accumulator arrangement to determine the long term desired configuration.1_/_

The cabling contacted by the EHC fluid will be inspected during the next refueling outage to determine if any deterioration is evident.1_/_

VII. ADDITIONAL INFORMATION

A. Failed Components:

None.

B. Previous Similar Events:

None.

C. Additional Information:

This event did not result in loss of the normal heat removal path as described in draft NEI 99-02, Rev. C, since the condenser was used for decay heat removal.

D. Safety System Functional Failure:

This event did not result in a safety system functional failure in accordance with draft NEI 99-02, Rev. C.

VIII. COMMITMENTS

None.

1_/_TVA does not consider this corrective action a regulatory commitment. The completion of this item will be tracked in TVA's Corrective Action Program.

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